Duke Energy®

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March 21, 2011

U.S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, D.C. 20555

Subject: Duke Energy Carolinas, LLC
McGuire Nuclear Station, Unit 1 and Unit 2
Docket Nos. 50-369 and 50-370
Licensee Event Report 369/2011-02, Revision 0
Problem Investigation Process Number M-11-00389

Pursuant to 10 CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report (LER) 369/2011-02, Revision 0, regarding a Unit 1 manual reactor trip and the completion of both Unit 1 and Unit 2 shutdown to MODE 3.

This report is being submitted for Unit 1 in accordance with 10 CFR 50.73 (a) (2) (iv) (A), "Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a) (2) (iv) (B)." The 10 CFR 50.73 (a) (2) (iv) (B) systems to which the requirements of paragraph (a) (2) (iv) (A) applied was the Reactor Protection System and the Auxiliary Feedwater System.

Since both Unit 1 and Unit 2 were in TS 3.0.3 at the time of the trip, Unit 1 entered MODE 3 and is reportable per 10 CFR 50.73 (a) (2) (i) (A), "The completion of any nuclear plant shutdown required by the plant's Technical Specifications." Unit 2 was subsequently placed in MODE 3 and is also reportable per 10 CFR 50.73 (a) (2) (i) (A).

The initiation of the dual unit shutdown due to the inoperability of all four trains of the Nuclear Service Water System is being reported separately under McGuire License Event Report (LER) 369/2011-01.

This event is considered to be of no significance with respect to the health and safety of the public. There are no regulatory commitments contained in this LER.

IE22
NRR

If questions arise regarding this LER, contact Rick E. Abbott at 980-875-4685.

Sincerely,

Regis T. Repko

Attachment

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NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION (10-2010)	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 08/31/2013
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA/Privacy Section (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
McGuire Nuclear Station, Unit 1	05000- 0369	1 OF 7

4. TITLE

Completion of Dual Unit Shutdown and occurrence of Unit 1 Reactor Trip.

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									McGuire Unit 2	05000 370
1	20	2011	2011	– 02	– 00	3	21	2011	None	05000

9. OPERATING MODE: Power Operation (MODE 1)

10. POWER LEVEL: 28%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

	20.2201(b)		20.2203(a)(3)(i)		50.73(a)(2)(i)(C)		50.73(a)(2)(vii)
	20.2201(d)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(A)
	20.2203(a)(1)		20.2203(a)(4)		50.73(a)(2)(ii)(B)		50.73(a)(2)(viii)(B)
	20.2203(a)(2)(i)		50.36(c)(1)(i)(A)		50.73(a)(2)(iii)		50.73(a)(2)(ix)(A)
	20.2203(a)(2)(ii)		50.36(c)(1)(ii)(A)	X	50.73(a)(2)(iv)(A)		50.73(a)(2)(x)
	20.2203(a)(2)(iii)		50.36(c)(2)		50.73(a)(2)(v)(A)		73.71(a)(4)
	20.2203(a)(2)(iv)		50.46(a)(3)(ii)		50.73(a)(2)(v)(B)		73.71(a)(5)
	20.2203(a)(2)(v)	X	50.73(a)(2)(i)(A)		50.73(a)(2)(v)(C)		OTHER
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(B)		50.73(a)(2)(v)(D)		Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Rick E. Abbott, Regulatory Compliance	(980) 875-4685

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On January 20, 2011, while shutting down Unit 1 and Unit 2, operators experienced a loss of feed water (CF) to the Unit 1 Steam Generators. At the time of the event, Unit 1 was at 28 % power and the operating crew was transferring steam supplies for the Unit 1 "B" (1B) CF Pump Turbine (CFPT) from the normal steam supply to the alternate steam supply. This required opening valve 1HM-95 to supply steam to the 1B CFPT while monitoring several parameters. Upon opening 1HM-95, the 1B CF pump discharge pressure increased to the trip set point and 1B CF pump tripped on high discharge pressure resulting in a loss of all CF flow. This was followed by an automatic main turbine trip and a manual trip of the reactor.

The root cause for the CF pump trip is the use of equipment for a purpose it was not designed. Transferring steam supplies to the 1B CFPT uses a gate valve (1HM-95) which is not designed for throttling. Actions will be taken to discontinue the use of 1HM-95 as a throttle valve and utilize the new CFPT governor steam supply controls for governor modulation.

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

BACKGROUND:

The following information is provided to assist readers in understanding the event described in this LER. Applicable Energy Industry Identification [EIIS] system and component codes are enclosed within brackets. McGuire Nuclear Station unique system and component identifiers are contained within parentheses.

Reactor Protection System [JC] (IPE):

The Reactor Protection System keeps the Reactor operating within a safe operating range by automatically shutting down the Reactor whenever the limits of the operating range are approached by monitoring process variables. Whenever a direct or calculated process variable exceeds a setpoint the Reactor is automatically tripped to protect against fuel cladding damage or loss of Reactor Coolant System (NC) integrity. Station operators may elect to manually actuate the reactor trip switchgear (manual reactor trip) using either of two control board switches.

Feedwater System [SJ] (CF):

The CF System takes treated Condensate [KA] (CM) System water, heats it further to improve the plant’s thermal cycle efficiency, and delivers it at the required flow rate, pressure and temperature to the steam generators. The CF System is designed to maintain proper Steam Generator (S/G) water levels with respect to reactor power output and turbine steam requirements. Those portions of the CF System that are safety-related perform the following functions:

1. Provide feedwater isolation to prevent overheating of the Containment in the event of a feedwater or main-steam pipe rupture in Containment.
2. Provide feedwater isolation or regulate feedwater flow to assist in mitigating the consequences of an inadvertent increase in heat removal by the secondary system.
3. Provide feedwater isolation to prevent flooding safety-related equipment essential to the safe shutdown of the plant in the event of a postulated pipe break in the main feedwater piping outside Containment.
4. Provide isolation of the steam generators to preclude a blow down of more than one steam generator following a feedwater line break.
5. Ensure containment integrity is maintained as required.

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

Auxiliary Feedwater System [BA] (CA):

The CA System automatically supplies feedwater to the steam generators to remove decay heat from the NC System upon the loss of normal feedwater supply. The CA System mitigates the consequences of any event with loss of normal feedwater. The design basis of the CA System is to supply water to the S/Gs to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators.

The motor driven CA pumps will start automatically and provide flow from the normal suction source within one minute on any one of the following conditions:

1. Two out of four low-low level alarms in any S/G.
2. Loss of all CF pumps.
3. Initiation of the safety injection signal.
4. Loss of off-site and station normal auxiliary power (blackout).

Main Steam System [SB](SM):

The SM System delivers the steam from the outlet of the S/Gs to the various system components throughout the Turbine Building. Steam is generated at essentially dry and saturated conditions. The SM System is designed to achieve the following:

1. Provide steam flow requirements at Turbine inlet design conditions.
2. Dissipate heat from the NC System following a Turbine and/or Reactor trip.
3. Provide steam for:
    a. CF and CA pump turbines.
    b. Condenser steam air ejectors.
    c. Main Turbine and CF pump turbine seals.
    d. Miscellaneous auxiliary equipment.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
McGuire Nuclear Station, Unit 1	05000369	2011	- 002 -	00	4 OF 7

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

EVENT DESCRIPTION:

Prior to the Unit 1 reactor trip, the Unit 1 and Unit 2 Nuclear Service Water System (RN) "A" and "B" Trains had been declared inoperable due to RN strainer macro-fouling. LCO 3.0.3 required that Unit 1 and Unit 2 be placed in MODE 3 within 7 hours (1/20/11 at 1810 hours); MODE 4 within 13 hours (1/21/11 at 0010 hours) and MODE 5 within 37 hours (1/22/11 at 0010 hours). The initiation of a TS 3.0.3 required shutdown was reported, as required, under 10 CFR50.72 (b) (2) (i), "The initiation of any nuclear plant shutdown required by the plant's Technical Specifications." The corresponding reporting criteria, 10.73(a)(2)(i)(B) to report an operation prohibited by Technical Specifications, is being reported in a separate LER (369/2011-01).

On January 20, 2011 at 1517 hours, while shutting down in accordance with TS 3.0.3, Unit 1 experienced a loss of feed water (CF) to the steam generators at about 28% power. The Unit 1 "A" CF pump was previously shut down in accordance with station procedures and the operating crew was transferring steam supplies for the Unit 1 "B" CF Pump Turbine (CFPT) from the normal steam supply to the alternate steam supply (Auxiliary Steam). This required the Reactor Operator (RO) to slowly open valve 1HM-95 while monitoring the Auxiliary Steam (AS) header pressure, 1B CF pump speed, S/G levels and CF flow. Upon opening 1HM-95, the RO noticed a small change in AS header pressure and 1B CF pump speed increased. Then, 1B CF pump discharge pressure rapidly increased until the 1B CF pump tripped on high discharge pressure. The 1B CF pump trip resulted in a total loss of CF flow, automatic turbine trip and automatic start of the motor driven CA pumps to supply feed water to the S/Gs. Operators then manually tripped the reactor in accordance with procedures.

The manual reactor trip placed Unit 1 in MODE 3 and, since the unit shut down was required by TS 3.0.3, this constitutes "The completion of any nuclear plant shutdown required by the plant's Technical Specifications" and is being reported per 50.73(a)(2)(i)(A). Additionally, on January 20, 2011 at 1801 hours, Unit 2 entered MODE 3 in accordance with TS 3.0.3 and station procedures. Since Unit 2 entered MODE 3 as required by TS 3.0.3, the completion of the Unit 2 shutdown is also being reported in accordance with 50.73(a)(2)(i)(A).

The Unit 1 manual reactor trip was initially planned as part of the required shutdown however, the timing and circumstances (loss of CF) were not pre-planned. This event was initially reported, as required, under 50.72

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

(b)(2)(iv)(B), "Any event or condition that results in actuation of the Reactor Protection System (RPS) when the reactor is critical except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation." This LER will satisfy the corresponding reporting criteria 10 CFR 50.73 (a)(2)(iv)(A), "Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B)." The applicable 10 CFR 50.73 (a)(2)(iv)(B) systems include the Reactor Protection System (RPS) and the Auxiliary Feedwater System (CA).

The relevant sequence of events is as follows (all times approximate, where date and hours are the same the event or action is occurring within seconds):

- 1/20/2011 @ 1512 hours: Unit 1A CF pump removed from service in accordance with station procedures.
- 1/20/2011 @ 1517 hours: Valve 1HM-95 opened to supply auxiliary steam to 1B CF pump.
- 1/20/2011 @ 1517 hours: 1B CF pump tripped on high discharge pressure.
- 1/20/2011 @ 1517 hours: Turbine trip on loss of both CF pumps.
- 1/20/2011 @ 1517 hours: 1B CA pump auto-start.
- 1/20/2011 @ 1517 hours: 1A CA pump auto-start.
- 1/20/2011 @ 1517 hours: Unit 1 Manual Reactor Trip A.
- 1/20/2011 @ 1517 hours: Unit 1 Manual Reactor Trip B.
- 1/20/2011 @ 1517 hours: Unit 1 Reactor Trip Switchgear A Tripped.
- 1/20/2011 @ 1517 hours: Unit 1 Reactor Trip Switchgear B Tripped.
- 1/20/2011 @ 1801 hours: Unit 2 was manually shutdown to MODE 3.

CAUSAL FACTORS:

The root cause for the 1B CF pump trip was the use of equipment for a purpose it was not designed. Specifically, the use of the gate valve for throttling purposes. This placed reliance on operator actions and procedure guidance to control critical operating parameters while transferring steam supplies to the 1B CF pump.

The use of a gate valve (1HM-95) to throttle steam supply to 1B CFPT required detailed procedures and proficiency to be successful. Contributing

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

to this event was inadequate procedural guidance to perform the evolution successfully. Procedure guidance was not sufficient for users performing the task without prior operating experience.

CORRECTIVE ACTIONS

Immediate:

1. Unit 1 reactor was manually tripped upon Main Turbine trip and applicable reactor trip response procedure steps completed.

Subsequent:

1. An immediate communication was issued to Senior Reactor Operators and Reactor Operators describing operation of 1HM-95 and 2HM-95.

Planned:

1. Discontinue the use of 1&2HM-95 as a throttle valve during the transfer of steam supplies to the CFPTs. Instead, utilize the recently installed Distributed Control System (DCS) features to allow modulation of the CFPT high pressure and low pressure governor valves during the transfer of steam supplies.

SAFETY ANALYSIS

The core damage significance and large early release significance of this event has been evaluated quantitatively by considering the following:

- A loss of main feedwater initiating event
- Actual plant configuration and maintenance activities at the time of the trip

There are two dominant core damage sequence types. The first type involves the loss of main feedwater initiator, failure of the Reactor Protection System and failure of the Operators to manually initiate emergency boration for an anticipated transient without a scram (ATWS) mitigation.

LICENSEE EVENT REPORT (LER)
CONTINUATION SHEET

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
McGuire Nuclear Station, Unit 1	05000369	2011	- 002 -	00	7 OF 7

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

The second sequence type involves the loss of main feedwater initiator, failure of the operators to restart main feedwater, loss of emergency feedwater due to common cause failure of the cooling water source and failure of feed and bleed cooling.

The Conditional Core Damage Probability (CCDP) associated with this event is evaluated to be < 1E-06. The Conditional Large Early Release Probability (CLERP) associated with this event is evaluated to be < 1E-07.

Note that both CF pumps and all three CA pumps were available to feed S/Gs during this event. Therefore, this event is considered to be of no significance with respect to the health and safety of the public.

ADDITIONAL INFORMATION

This event, CA auto-start on a loss of Feedwater with a manual Reactor trip, was not considered to be recurring, although McGuire has experienced system perturbations and plant transients in the past using gate valve 1HM-95 for throttling.